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Abstract

STUDIES OF CANDU-TYPE REACTORS IN THE 500-MW(e) RANGE

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1. The D_2O-cooled CANDU-type reactor has exceptional potential for development to higher power and lower energy costs. The concept is clearly established with the operation of the 20-MW(e) NPD plant and the design and construction of the Douglas Point station. The power cost from the first 200-MW(e) unit at Douglas Point is expected to be approximately \$(Can) 5.5×10^{-3} /kWh, based on operation for 7000 hours per year and a fixed-charge rate of 6.5 percent. If a number of these 200-MW(e) plants were built in succession utilizing current designs and overlapping construction, power could be produced at a price fully competitive with other forms of generation in many areas in the world.

2. However it has been evident for some time that substantial savings in capital and power cost would result from building larger reactor plants. The pressure-tube reactor can be designed and built in any size acceptable to a utility. Increasing the size from 200 MW(e) to 500 MW(e) without any technological changes in conditions or materials would result in a capital-cost saving of about \$(Can)100 /kW(e) of installed capacity. The resulting power-cost reduction would be 20 percent. Substantial reductions in cost would result from further increases in reactor size to 1000 MW(e) - and perhaps beyond.

3. Power-cost reductions are also predicted from a number of technological improvements and by reductions in time of construction. Technical changes which could result in power-cost savings include increased fuel ratings, improved high-strength materials for pressure-tube construction, allowing the coolant to boil in the core, axial flattening of the neutron flux, and higher operating temperatures. These changes would lead to a smaller core and substantial savings in power cost. Savings in construction cost and time, and in capital and operating costs, would result from building a number of units in succession at one site.

4. The low fuelling cost which results from the use of natural-uranium fuel and D_2O moderation would prevent future obsolesence as larger advanced plants are built. There is promise of even lower fuelling cost with increased fuel fabrication throughput as the nuclear industry develops.

5. A design study for the construction of a 500-MW(e) reactor as the first unit of a multi-reactor generating station in southern Ontario is in progress and the conclusions reached can be reported in some detail by September, 1964.

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Abstract

MANUFACTURING AND CONSTRUCTION FOR THE DOUGLAS POINT NUCLEAR POWER PROJECT

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1. The Douglas Point Project, located on the east shore of Lake Huron, is scheduled for completion in 1964 and for the generation of 200 megawatts of electricity in 1965. Brief description is given of the conventional aspects of the plant and a more detailed description of the fabrication and construction of the CANDU reactor system which provides the energy.

2. The following topics are discussed:

- The manufacture of the more important reactor components.

- The development and application of special techniques and procedures used in the fabrication of the calandria, end shields, coolant assemblies, steam generators and associated equipment; especially tube rolling, tubesheet welding, hydraulic and mechanical forming, and special machining operations.

- Requirements for high precision in certain reactor components; the methods used to attain this precision in the construction of CANDU components.

- Site organization and management; reviewing the construction schedule and manpower requirements.

- Site and construction plant facilities.

- Methods used in the erection and construction of the reactor and the reactor building.

- The forming, mixing and placing of light and heavy concrete in the reactor building and vault structure; methods used to achieve

the required shielding densities, dimensional accuracy and to minimize the effects of shrinkage.

- The construction of the reactor vault, the erection of its thermal shields and the containment liner for its atmosphere.

- Details of handling, preparation and installation of reactor components on site; the provisions and methods for moving large components into the reactor vault and the installation and alignment procedures for the end-shield rings, the end shields, calandria and dump tank.

- Special methods used in the installation of the reactor-coolant assemblies and in the erection of piping and equipment for the primary heat-transport system.

- The use of mock-up structures to prove installation procedures and train working forces.

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Abstract

PROBLEMS AND PERFORMANCE OF NPD

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1. The NPD (Nuclear Power Demonstration) reactor is a 20-MW(e) prototype designed to show the technical feasibility of building and operating nuclear power stations with heavy-water moderator, heavy-water coolant, horizontal pressure tubes, refuelling at full power and a natural UO_2 fuel with no reprocessing. It is the forerunner for the CANDU reactor and other large units designed to produce power economically.

2. The NPD reactor went critical in April 1962, and reached full power in June 1962. Since October 1962 the time has been divided between "Capacity Runs" and "Improvement/Test Periods".

3. The "Capacity Runs" correspond to normal operation of a base-load station, and are intended to prove reliability (high capacity factor) and to identify deficiencies. During the "Improvement/Test Periods", improvements are made to eliminate the deficiencies, and special tests and inspections associated with the pioneer nature of NPD are carried out.

4. In three "Capacity Runs", totalling 29 weeks, the capacity factor has averaged 78%. The major causes of lost production have been eliminated and capacity factors of 90% are anticipated in future. "Capacity Runs" lasting 12 months are planned.

FUEL

5. None of the fuel has failed. By December 1, 1963, the centre fuel had reached 3800 MWd/tonne and average burn-up was 1300 MWd/tonne. Design rating of the fuel is 6500 MWd/tonne but some fuel will be exposed much longer.

ON-POWER FUELLING

6. The fuelling machines have performed over 1500 operations at low power and low temperature. The first attempt at high power on December 3, 1962, was a failure. A large spill of heavy water occurred when the seal between one fuelling machine and the reactor failed.

7. The sealing mechanism has been improved and instruments have been developed to measure the seal performance.

8. On November 24, 1963, on-power fuelling became a reality and NPD became the first high-pressure reactor with this ability.

PRESSURE TUBES

9. An in-reactor inspection of one of the Zircaloy pressure tubes was carried out in October, 1963, after 17 months of operation. The tube surface was in excellent condition, with negligible damage due to fuel movement. Another inspection is planned in October, 1964, and later a tube may be removed for more detailed examination and testing.

PRIMARY-PUMP SEALS

10. The main D_2O coolant pumps are standard pumps with mechanical shaft seals. Long seal life is vital. At first, dissolved gas in the coolant caused seals to fail after a few hundred hours. Since then, dissolved gas has been greatly reduced and seal venting has been improved. As a result, seals recently inspected after 3000 hours are in excellent condition.

STATION CONTROL

11. Reactor control and response to disturbances have been excellent.

D_2O LOSSES

12. These are still somewhat above pre-operation estimates. Sophisticated leak-detection techniques are being used to reduce them.

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Abstract

NPD START-UP, REACTIVITY BALANCE AND FUEL MANAGEMENT

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1. NPD is a 20-MW(e) power reactor, and is moderated and cooled by heavy water. It has horizontal pressure tubes containing nine 19-element UO_2 fuel bundles in each of 52 central channels, and nine 7-element bundles in each of 80 outer channels. As there are no absorber rods for reactivity control, and as full power cannot be obtained at low moderator level, 62 of the 19-element bundles were partially depleted in the fissile isotope to reduce the initial excess reactivity.

2. With all fuel in position, criticality was approached by raising the moderator level, and the initial critical height was compared with that calculated by two-group neutron-diffusion theory. The reactivity effect as the moderator level rises from the initial critical height to the top of the tank was measured by adding known amounts of $CdSO_4$ to the coolant; the total observed change in the effective multiplication constant, k_{eff}, was 0.04 and this agrees with the calculated value for the cold, clean reactor. The moderator-temperature coefficient for k_{eff} between 21°C and 74°C was found to vary from -125×10^{-6} $degC^{-1}$ at minimum moderator height to -74×10^{-6} $degC^{-1}$ at maximum height, with fresh fuel. The coolant-temperature coefficient was -36×10^{-6} $degC^{-1}$.

3. When the reactor was first operated at high power, the power coefficient of k_{eff} was found to be -31×10^{-6} per 1% change from full power. Further measurements of the power and temperature coefficients during the first 220 full-power days indicate that all three coefficients became significantly less negative with the build-up of plutonium in the core. Excluding xenon and samarium effects, the change in k_{eff} on going from zero to full power was observed to be -0.013 with fresh fuel, compared to an estimated -0.004 at equilibrium. The xenon and samarium loads at equilibrium irradiation are calculated to be 0.024 and 0.009

respectively. The initial excess reactivity available for fuel burn-up was 0.023 in k_{eff}.

4. Other measurements in NPD include the reactivity worths of the ^{235}U-Al 'booster rod' and the H_2O reflector. Radial flux distributions were measured, and compared with two-group diffusion calculations. The photoneutron source measured after an extended shutdown was in satisfactory agreement with calculation. Measurements of the reactivity effect of H_2O in D_2O moderator also confirmed calculated values.

5. The reactor is designed to operate continuously at high power, without having to shut down for fuel shifts. Principles of fuel management, for controlling both the flux distribution and the reactivity, have been determined by computer simulation of reactor behaviour under various fuelling schemes. The two-dimensional program, STOKE, for this study assumes cylindrical symmetry, but allows for the fuelling of alternate channels from opposite ends of the reactor. A three-dimensional program, PERIGEE, has been used to simulate the operating history of the reactor to date, taking changes in moderator level and purity into account as well as all fuel shifts. The calculated distribution of power among the channels agrees with the measured distribution, within the accuracy of the reactor instruments. STOKE and PERIGEE have also been used to estimate equilibrium fuel burn-up, giving an average exit irradiation exceeding 6000 MWd/tonne U.

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Abstract

PROSPECTIVE D_2O-MODERATED POWER REACTORS

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1. Capital-cost estimates for very high power reactors of several hundred up to 1000 megawatts (electrical) indicate that many types would be competitive. In each case, however, an optimum technical design must be sought. Cool heavy water offers several advantages as moderator; not only does it improve neutron economy but it also aids containment and pressure suppression in the event of a major leak of high-temperature water or steam. The hot fuel channel may be insulated from the cool moderator by a gas gap between a hot pressure tube and a cold "calandria" tube. Alternatively, the stress tube may be cooled and carry solid internal insulation with a thin metal liner. Zirconium alloys have proved supreme for these tubes and may be further developed for strength and corrosion resistance. One possible heat-transport fluid or caloporteur would be a radiation-resistant organic liquid, such as terphenyl. For this fluid aluminum or SAP (an aluminum-alumina sinter) offer some advantages. Liquid D_2O is an established caloporteur of continuing promise; also either D_2O or H_2O may be used as a two-phase steam-liquid mixture. A minimum volume of caloporteur is required for several reasons: to preserve neutron economy, minimize reactivity changes arising from density changes or voiding, and to minimize the stored energy and in the case of the organic to minimize radiolytic decomposition.

2. In the long-term future, most fission energy from thermal-neutron reactors seems likely to be derived from thorium via uranium-233. For the next several decades, however, natural uranium appears most promising and economical. Enriched uranium has been included in the studies.

3. Preselection of reactor types led to detailed optimization and costing evaluations of reactors of large capacity using the caloporteurs mentioned. Fuels of various configurations and, in management, schemes allowing axial and radial flux flattening were evaluated. UO_2, UC and possible U metal alloys were considered where applicable.

4. The general method of assessment was to establish an electronic-computer program in which nuclear data, materials properties and costing information would be combined to yield optimum costs without transgressing selected limits of flow velocities, temperatures, stresses and other conditions. Several hundred reactors have been evaluated in this manner. From these results promising "reference" designs were chosen for each selected caloporteur. Preliminary mechanical designs were made for each reference case, and capital-cost estimates were made from these designs and fed back into the power-cost evaluation.

5. The general conclusion is that the greatest power-cost reductions will come from increasing the capacity of each reactor and of the complete plant. Beyond this, the major cost reductions follow from improved neutron economy by the use of thin-walled tubing and of dense fuel and from improved thermal-to-electric conversion efficiency by raising the mean temperature of the caloporteur. Significant reductions would also result from the expected further lowering of the cost of heavy water, and from improved construction methods. The use of thorium fuel does not appear favourable unless ^{235}U becomes available at less than \$12/g or the cost of natural uranium rises above about \$40/kg U.

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Abstract

ANALYSIS OF CAPITAL COSTS FOR NUCLEAR POWER PLANTS

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1. This paper reviews a capital-cost estimate for a typical 200-MW(e) heavy-water-cooled and -moderated nuclear power plant designed for construction in Canada. The system of accounts developed by Atomic Energy of Canada Limited and Canadian General Electric Company Limited for CANDU and NPD is described with particular emphasis on the type of costs included in the indirect charges. The estimate is broken down to show costs incurred because of local safety regulations, equipment or technique development of non-standard components and site labour; and capital-cost reductions for future nuclear plants are indicated.

2. Changes in investment-capital requirements if the nuclear plant were sold by Canadian industry for installation in an industrial foreign country - and alternatively in a developing foreign country - are estimated and the foreign and domestic capital contents are analyzed. Typical construction schedules and cash-flow analyses are presented as well as the effects of foreign exchange and domestic and foreign purchases on the investment-capital requirements. The effects of national architectural principles, local design requirements and site costs on the design and estimate are discussed, and a bill of material for the buildings of a typical plant located in Canada is illustrated. Site-labour costs for mechanized construction industries in Canada and Europe are compared with the lower labour costs of the developing countries.

3. Separation of the responsibilities of the seller, buyer and governments in the establishment and funding of the nuclear manpower training program is attempted.

4. The relative effect of plant size on certain cost components is illustrated for a range of outputs from 100 MW(e) to 500 MW(e).

5. The advantages accruing to an industry in subsidizing components for the initial plant are pointed out together with the danger of estimating the cost of future plants from that of the initial plant.

6. The desirability of common ground rules in the preparation of capital-cost estimates in the indirect-cost area is discussed.

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Abstract

RESEARCH WITH INELASTIC NEUTRON SCATTERING AT THE NRU REACTOR

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1. Four spectrometers have been in use at the NRU reactor for several years for the study of liquid- and solid-state phenomena by inelastic neutron scattering. Two are crystal spectrometers which define the energies of the incoming and scattered neutrons by Bragg reflection from single crystals. The other two use pulsed beams and time-of-flight techniques.

2. The triple-axis crystal spectrometer allows automatic variation of incoming and outgoing neutron energies, angle of scattering, and specimen orientation. With this spectrometer it is possible to measure the energy distribution of scattered neutrons while maintaining the (vector) momentum transfer constant. A second crystal spectrometer, designed by D.G. Henshaw for investigations of liquid helium, combines such features as a 400-cm^2 beam, high resolution, and low background conditions.

3. The time-of-flight machines are of two types. The first (designed by P.A. Egelstaff) employs a high-precision phased-chopper system to give a pulsed monoenergetic beam of neutrons. The energies of the scattered neutrons are determined from their times of flight to one of many scintillation detectors covering a large range of scattering angles. The second time-of-flight machine uses a spinning single crystal to give the monoenergetic incident neutrons; it has mostly been used for work at long wavelengths and high resolution.

4. Four principal fields of study have been explored using these instruments:

- lattice vibrations in solids,
- atomic and molecular motions in liquids,
- spin-wave spectra in magnetic materials,
- inelastic neutron scattering cross-sections of interest for reactor-spectra calculations.

5. The investigations of crystal dynamics have been principally concerned with determination of the frequency/wave-vector dispersion relation $\nu(\vec{q})$ for the normal modes of vibration. Among the substances so studied are metals (lead, sodium, niobium, molybdenum, tantalum and tungsten), an alloy (β-brass), semiconductors (germanium, silicon and gallium arsenide), insulators (sodium iodide and potassium bromide) and miscellaneous materials such as pyrolytic graphite and strontium titanate.

6. Extensive experiments on atomic motions in water were made at several different temperatures. Liquid bromine, liquid lead, liquid tin, liquid argon and liquid helium have also been investigated; the experiments on helium were carried out with high resolution for the purpose of determining accurately the shape of the Landau excitation curve below the λ transition temperature, whereas the experiments on lead and bromine were carried out over a large range of energy and momentum transfers to obtain a general picture of the molecular motions.

7. The experiments on magnetic systems have been chiefly concerned with the determination of the energy/wavelength dependence of spin waves in cobalt (with 8% iron) and in magnetite. Other experiments have been designed to obtain information about specific magnetic levels in materials such as ytterbium iron garnet and rare-earth oxides.

8. Inelastic scattering cross-sections have been measured over a wide range of energy and momentum transfer, and attempts have been made to deduce the main features of the frequency-distribution function $g(\nu)$ from the results. Beryllium, beryllium oxide, graphite, uranium oxide and many hydrogenous compounds are among the solids studied.

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Abstract

STUDIES OF THE FISSION PROCESS INDUCED BY THERMAL NEUTRONS

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1. With a high-intensity neutron beam available from the NRU reactor, the prompt mass and kinetic-energy distributions resulting from the thermal-neutron fission of ^{233}U, ^{235}U and ^{239}Pu have been re-examined using the time-of-flight technique to measure simultaneously the velocities of the fragment pairs. With the high resolution now available a fine structure in the mass yields is observed. This structure becomes more pronounced at high total kinetic energies where the excitation energy is low. There is a strong correlation between the positions of the maxima in the mass yields and maxima in the energy release for even-even fragments. A similar fine structure in the spontaneous fission of ^{252}Cf has been shown to be associated with the same heavy even-even fragments.

2. The large decrease (35 MeV) in the total kinetic energy for nearly symmetric mass divisions has been shown to be connected with an increase in the total yield of prompt neutrons. This has been done for a ^{233}U target in an experiment to measure the correlation in energy and angle of the prompt neutrons with fission-fragment mass and energy. Neutron flight paths of 1 meter were set up at angles of 10°, 25°, 45° and 80° to the fragment flight paths. Because the background is relatively high at the larger angles, only the data taken at 10° has yet been analyzed in detail.

3. The average number, ν, of neutrons emitted by a fragment varies with fragment mass and is in excellent agreement with the indirect determination made by Terrell from a comparison of prompt and final mass yields. ν is nearly zero at mass 80, rises to a peak near mass 110, falls to zero near mass 130, then rises monotonically through the heavy-fragment group.

4. In contrast to the sawtooth behaviour of ν, the variation of average neutron kinetic energy is symmetrical about the mass symmetry point. If the neutron energy is taken as an indicator of

the temperature of the fragment, it follows that the temperature of a mass 132 fragment in the neighbourhood of the neutron binding energy is equal to that of a mass 102 fragment some 25 MeV higher.

5. These observations suggest that there is a great dissimilarity in the stiffness of the two fragments and that they are highly deformed with low temperatures at the point of scission, the deformation energy being converted to excitation energy at a later stage. The observed variation of ν with the total internal energy of the fragments supports this view.

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Abstract

THE INSTALLATION, OPERATION AND MAINTENANCE OF EXPERIMENTAL LOOPS IN REACTORS

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1. Reactor loops have been used at Chalk River since 1950 to study the behaviour of fuel and coolant systems under the thermodynamic and nuclear conditions that hold in power reactors. There are now eight such loops in the NRX reactor and fuel can be tested at high heat ratings in pressurized water, fog, steam and organic coolant. The NRX loops are limited mainly in the size of the specimens that they will carry and the neutron fluxes that are available. To overcome these limitations the NRU reactor has been equipped with loops large enough to test full-size fuel assemblies. One loop operates with water coolant in either one- or two-phase condition; another operates with organic coolant; the third is a water-steam loop that has been designed and will be installed in 1964.

2. One of the NRU loops is in the central vertical core position and the others are at sites 79 cm from the centre. Depending on the loading the thermal-neutron flux in this region ranges up to $3.10^{14}\,cm^{-2}\,s^{-1}$ and the heat output from a typical irradiation is 2 to 2.5 MW. The large size and complexity of such systems introduced many new problems in design, operation and maintenance.

3. A primary consideration is efficient layout so that components can be erected in limited space. Ready accessibility for maintenance and modification is essential. Each in-core position is fitted with highly stressed large-diameter zirconium-alloy pressure tubes. Reliable closures and joints were designed for these tubes and suitable methods developed to protect them against overpressure.

4. Operating a large organic-cooled loop at 400°C and 32 bar requires many new techniques. The relatively high pressure demands mechanical seals of high integrity with extremely reliable cooling. Thermal insulation, steam tracing, fire protection and coolant degassing presented problems in both operation and maintenance.

5. Experience in operating the small boiling-water and steam loops in NRX provided useful guides in designing the large, multipurpose, water-steam loop for NRU. This unit will be used over such a wide range of operating conditions that comparatively sophisticated instrumentation is needed for stable control.

6. Loop experiments produce large volumes of experimental data. The usefulness of automatic data acquisition and processing in situ has been demonstrated on a pilot scale and all Chalk River loops will be serviced by a combination of data logger and computer in 1964.

7. Chalk River loop experiments have led to the design and production of fuel that has performed successfully in the NPD reactor and to the production of low-cost fuel for the CANDU reactor. The loop program will be continued and expanded for studies of improved fuel systems. A loop is being designed and built for the Canada-India Reactor at Bombay; and Canada's next test reactor WR-1 will test fuel at power-reactor conditions in many of its channels.

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Abstract

DEVELOPMENT OF ORGANIC LIQUID COOLANTS

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1. Research and development work, mainly in fuel-test loops in the NRX and NRU reactors, shows that organic liquids are promising media for transporting heat from reactors, that their purity can be maintained in the presence of radiation and heat fluxes, and that structural materials are available that are compatible with these liquids.

2. Thermal and radiolytic decomposition rates of individual terphenyls and of mixtures have been measured over the temperature range from 150 to 475°C. For the same energy deposition, fast neutrons cause about three times as much damage as electrons. Decomposition rates have also been measured for other liquids which have melting-point properties suitable for initial reactor start-up. One, a partially hydrogenated terphenyl mixture, had a decomposition rate about the same as that of the terphenyls but a gas-formation rate about three times higher.

3. Continuous purification of the liquid is essential. Chlorine in the terphenyl used for make-up varies from 2 to 10 ppm and the amount in the bulk system must be kept below about 5 ppm. Chlorine can be removed on beds of magnesium turnings or of palladium on alumina granules heated above 325°C; in routine operation of loops the level can be kept below 3 ppm. Other impurities are removed by adsorption on beds of attapulgus clay at 325°C. Particulate matter must be kept to a minimum by filtration, e.g., through glass-wool filters. Water and hydrogen can be removed by gas stripping, and, while the hydrogen should be kept low, reasons exist for controlling the water at a level around 100 ppm.

4. Two types of fouling on fuel-element surfaces have been recognized. The first type gives a deposit containing over 70% iron with essentially no organic matter. It is formed by the transfer of iron from the circuit piping at about 300°C to the hot fuel-element surface at 400 - 500°C. Chlorine, most of which resides on the system walls, plays an important role in the transfer. If the total chlorine in the system is less than the equivalent of 3 ppm in the coolant, this mass-transfer effect is acceptably low.

The second type of fouling is caused by organic particles in the coolant and gives a deposit high in organic matter. High surface temperatures increase the amount of this type of fouling, but irradiation appears not to influence it greatly. Recent experience in the X-7 loop in the NRX reactor showed that after 82 days irradiation in a thermal-neutron flux of about 5×10^{13} cm^{-2} s^{-1}, the fouling film on fuel sheaths maintained at 450 to 500°C was less than 1 μm thick.

5. Zirconium alloys are shown to be compatible with organic coolants provided the coolant conditions are correct. Chlorine promotes rapid hydriding, but when the chlorine content is kept below 3 ppm the effect is markedly decreased. The concentration of water is also very important, since, if it is too low, an oxide film impermeable to hydrogen is not maintained; the optimum water concentration is probably about 100 ppm. The longest test was of a Zircaloy-4 pressure tube in the X-7 loop. After 10 months of irradiation with a coolant outlet temperature of 350 - 390°C and a low water concentration (10 - 50 ppm) the maximum hydrogen pick-up by the alloy was about 80 ppm. There was some evidence of higher pick-up in regions of higher flux.

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Abstract

THERMAL AND IRRADIATION PERFORMANCE OF EXPERIMENTAL FUELS OPERATING IN STEAM-WATER MIXTURES

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1. Steam-water mixtures (fog) are capable of transporting large quantities of heat from a reactor; yet they do not demand excessive pumping power and do not introduce bulk quantities of neutron-absorbing material into the reactor core. Before fog is used in power reactors, more understanding must be obtained by research and development in three areas:

- thermal and hydrodynamic properties of fog systems
- fuel-sheathing materials suitable for use in fog
- chemistry of fog systems to minimize corrosion and deposition.

2. Electrically heated elements were used to simulate fuel in laboratory experiments carried out to

- measure "dry-out" heat fluxes for fuel strings ("dry-out" is the heat flux at which the film of water disappears from the heat-transfer surface and the heat-transfer coefficient suddenly falls)
- determine heat-transfer coefficients at heat fluxes beyond dry-out
- determine the effect of having elements of different lengths and having sections heated and unheated.

3. Five irradiations of fog-cooled fuel have been carried out in a loop in the NRX reactor. In the first irradiation the onset of dry-out was measured and it was shown that surface thermocouples could indicate incipient dry-out.

4. The second irradiation was run for 75 days, and provided data on the corrosion and hydrogen pick-up of fully wetted fuel sheaths fabricated from two different zirconium-base alloys.

5. The next three irradiations were undertaken to determine the corrosion and hydriding rates of zirconium alloys when operating beyond

dry-out at heat fluxes of 60, 70 and 90 W/cm^2. The time-to-failure at the highest of these heat fluxes was determined to be approximately 30 h. This compares reasonably well with predictions obtained from out-reactor corrosion rates with an allowance for the insulation of the corrosion film. The formation of defects was found to be gradual and did not result in catastrophic failures. Sheath temperatures up to 670° C were encountered for extended periods during these tests. In each of the three irradiations in this group, some of the fuel elements in the string had dry surfaces and the remainder had wetted surfaces. The transition from wet to dry surfaces was assisted by disrupting the flow of the water film with a surface discontinuity. The tests indicated a possible problem from thermal distortion of a fuel element when only part of its sheath is dry; this can accelerate the time to failure.

6. The irradiation tests have covered a range from 27 to 63 wt% steam at the outlet, and mass velocities from 107 to 136 g cm^{-2} s^{-1}; all the tests were conducted at a system pressure of 70 kgf/cm^2. In general, the dry-out results obtained from the irradiation experiments and the laboratory tests agree well with one another, and with those predicted from calculations of the flow of liquid films on heat-transfer surfaces.

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Abstract

UO_2 PERFORMANCE - THE IMPORTANCE OF TEMPERATURE DISTRIBUTION

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1. The performance of UO_2 fuel elements is largely determined by the temperature distribution in the fuel. On this factor depend the disposition of the fuel, its expansion behaviour and the pressure developed by released fission-product gases.

2. Irradiated UO_2 pellets frequently exhibit cracks induced by thermal stresses. Pellets irradiated with preformed cracks demonstrated that healing occurs out to a radius slightly greater than that for just-discernible grain growth in the UO_2. Within this region large cavities closed during irradiation, but they remained unchanged in the outer annulus. Tungsten markers have confirmed that columnar grains can form below the melting point. Those columnar grains that are formed by freezing are identified by their characteristic structures of sub-grain boundaries and pores.

3. From 400 to 1600°C the mean value for the thermal conductivity of sintered UO_2 during irradiation is 0.03 $W\,cm^{-1}\,degC^{-1}$, deduced from thermocoupled elements and observations of grain growth. At higher temperatures, the interpretations of melting lead to a mean value of 0.035 $W\,cm^{-1}\,degC^{-1}$ affected little, if any, by grain growth. Following an irradiation demonstrating the relatively poor thermal conductivity of UO_{2+x}, simple comparative tests in the laboratory suggested that UO_{2-x} has improved conductivity. In a subsequent irradiation the central temperatures in $UO_{1.98}$ specimens were as much as 100 degC lower than those of $UO_{2.007}$ controls at 1200°C. Potentials measured between the fuel centre and surface showed that hypostoichiometric urania is an n-type conductor (hyperstoichiometric urania is p-type).

4. Post-irradiation collection of the released fission-product gases, supplemented by analyses for radial distribution of the gas remaining, confirmed that the overall release process is thermally activated. Comparison of elements irradiated alongside each other has shown a progres-

sive decrease in release with increasing density in the range 10.4 to 10.75 g/cm^3. Measurements made during irradiation indicated that the gas pressure can decrease markedly with minor sheath strain due to fuel expansion.

5. Post-irradiation observations on elements containing internal gauges defined conditions for which a stack of cylindrical UO_2 pellets is restrained from expanding thermally into an end-clearance. Measurements during irradiation showed how restraint can cause plastic deformation of the UO_2 at temperatures over approximately 750°C. A "plastic-core" model for the fuel can also explain the diametral expansion of the UO_2, which is typically 2.2% of the diameter at the onset of central melting. Thereafter, an increase in the rate of diametral expansion with power output is attributable to the volume change on melting.

6. At the 1958 Geneva Conference, an $\int_{\text{surface}}^{\text{centre}} \lambda d\theta$ value of 40 W/cm (λ is the thermal conductivity of the fuel at temperature θ) was recommended for sintered UO_2 in water-cooled power-reactors. The acceptability of thin, collapsible Zircaloy sheaths was subsequently established in tests taken to 10 000 MWd/tonne U. Experiments with $\int \lambda d\theta$ values up to 85 W/cm have now indicated the feasibility of operating power-reactor fuel up to 50 W/cm.

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Abstract

ENGINEERING AND PERFORMANCE OF UO_2 FUEL ASSEMBLIES

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1. The prototype 20-MW(e) NPD and the full-scale 200-MW(e) CANDU are the first of a series of Canadian power reactors based on heavy-water moderation, heavy-water cooling and natural-uranium fuel. The key to low-cost power from reactors of the CANDU type lies with low fuelling costs. The purpose of this paper is to show how CANDU-type fuel has been engineered for burn-ups in excess of 10 000 MWd/tonne U to yield total fuelling costs of less than 0.8 mills/kWh. The following points are discussed:

NEUTRON ECONOMY IN DESIGN

2. The fuel is made up of elements 15.2 mm in diameter which are assembled in clusters of 19 to make "bundles" 49.5 cm long. Several bundles stacked end to end make up the fuel charge in one channel (82.5 mm I.D.). This design allows bidirectional fuelling and hence radial flux flattening and uniform burn-up of all the fuel in each channel.

3. The UO_2 is clad in Zircaloy-2 and a fuel/cladding weight ratio of 10 is achieved principally by designing for cladding (0.38 mm thick) that will collapse on to the fuel under the pressure of the coolant.

4. A detailed discussion is given of the problems associated with meeting the hydraulic, thermal, structural and dimensional limitations imposed by the requirements for compatibility with the Zircaloy pressure-tube assembly and with the on-power fuelling machine.

IRRADIATION PERFORMANCE

5. The CANDU and NPD programs have placed heavy emphasis on irradiation testing. Fifty-four experimental, full-scale prototype fuel assemblies have been irradiated at coolant temperature and pressure

covering a range of heat ratings ($\int \lambda d\theta$) from 20 to 55 W/cm (maximum nominal rating for CANDU is an $\int \lambda d\theta$ of 40 W/cm).

6. Fuel loading of NPD was completed in March 1962; since then the first charge has logged 210 equivalent-full-power days of operation, mostly at full rating. As yet none of the 1188 fuel bundles has failed. Sample bundles extracted from the highest-rated positions have been found to be in excellent condition.

FUEL COST

7. CANDU first charge has been purchased at a fixed price which assures fuelling costs of 1.1 mills/kWh ± 10%. Based on NPD and CANDU fuel manufacturing experience, estimates indicate that the expansion of throughput to meet a 1000-MW(e) capacity will bring fuelling costs down below 0.8 mill/kWh. This economy has been achieved by engineering the fuel design and fabrication methods to be mutually compatible. The development of resistance welding has been a major factor in reducing fabrication costs.

8. Emphasis has been placed on the evolution of product specifications.

FUTURE POTENTIAL

9. Experiments are underway to increase allowable heat rating and introduce boiling to improve channel output. Concurrently, innovations are being introduced to further reduce the Zircaloy inventory and simplify fabrication methods.

10. Preliminary experiments on full-scale prototype fuel bundles have been successfully completed with heat ratings 130% of those for CANDU, with boiling in the channel to give as much as 13wt% steam at the outlet, and to a calculated burn-up of 7000 MWd/tonne U.

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Abstract

FISSION-GAS BEHAVIOUR IN UO_2 FUEL

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1. It is known that relatively large bubbles appear and grow in UO_2 fuel at high irradiations. Designers of fuel for burn-up much above 20 000 MWd/tonne U (5×10^{20} fissions/cm^3) need to know how the fission gas will distribute between such bubbles, designed void spaces and the body of the oxide. Experiments to establish understanding of the phenomena are discussed.

2. These experiments confirmed by prototype fuel irradiation indicate that fission-gas release is not a problem with our present power reactor design. For some other designs it appears necessary to consider many of the observed phenomena of fission-gas migration. Experiments have shown that some of the fission gas will migrate to internal traps in the UO_2 rather than escape into void spaces of the fuel element. Other experiments have shown that considerable xenon is re-absorbed from the gas into the oxide. The overall behaviour is strongly dependent on temperature and thermal gradients as well as previous history, stoichiometry of the oxide, porosity, mechanical constraint, and the total irradiation.

3. The following observations from recent experiments are reviewed:

3.1 The apparent diffusion constant, obtained by annealing sintered UO_2 after irradiation, decreases with time in a way that suggests not only trapping of gas in closed pores but also some re-absorption from the pores.

3.2 After irradiated UO_2 had been annealed at 1700°C for 3 hours to allow fission xenon to diffuse, pulverizing released 55% of the total xenon.

3.3 The fraction of fission xenon that is free to diffuse in post-irradiation annealing is decreased by increasing the prior irradiation exposure.

3.4 The migration of lenticular "voids" up a thermal gradient, over the range 1700 to 2200°C, can be very rapid and is believed to sweep the fission gases towards the fuel centre. Fuel cracking during reactor shutdown would liberate the gas from any exposed "voids".

3.5 Very low irradiation at temperatures up to at least 250°C causes the attachment of xenon to UO_2 surfaces exposed to the gas. Much of this gas is released by heating to 400°C. Very little remains attached above 800°C.

3.6 Longer irradiation results also in much more firmly attached xenon that is not significantly released below 1200°C. The amount embedded is roughly proportional to the irradiation, gas pressure and exposed surface. Samples have contained amounts up to 3.6 times the total produced by fission at 2500 MWd/tonne U.

3.7 In one series of experiments considerable isotopic disproportionation was observed in the absorbed xenon. This appears attributable to several processes of migration through narrow channels and over surfaces.

3.8 Post-irradiation studies on several UO_2 rods showed the expected sigmoidal distribution of fission-product xenon from surface to centre. However, in another rod the lowest xenon concentration did not correspond with the highest temperature. Almost complete retention occurs in dense UO_2 below about 1000°C for irradiations up to 10 000 MWd/tonne U.

3.9 Observed burst releases of fission gas appear attributable, in association with surface trapping, to stoichiometric equilibration, evaporation along fission-fragment tracks and to a lesser extent to other phenomena such as nuclear recoil.

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Abstract

EXPERIENCE WITH THE CHEMISTRY OF WATER IN MODERATOR AND COOLANT SYSTEMS

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1. This paper summarizes information from the NRX, NRU and NPD reactors and from experiments in in-reactor loops cooled by pressurized water, boiling water or steam-water mixtures (fog).

RADIOLYTIC REACTIONS IN MODERATORS

2. The steady-state concentrations of dissolved deuterium formed by radiolytic decomposition vary from 1 to 5 cm^3 (STP)/kg D_2O. Temperature, pD and other undefined factors affect the radiolytic equilibria. The net rates of decomposition vary from ~0.4 to 30 liters (STP)D_2 per MWh of total reactor output and depend on the rate at which dissolved gas is transferred to the helium atmosphere over the moderator.

3. With an air-to-helium ratio of 1:1 in the atmosphere at the top of the NPD reactor vessel, about 1 g mole of nitric acid is produced per day. This rate depends linearly on the total volume of nitrogen gas present.

RADIOLYTIC REACTIONS IN COOLANTS

4. In pressurized-water systems a hydrogen or deuterium concentration of 5 cm^3/(kg water) is adequate to maintain oxygen below 0.01 cm^3/kg and to suppress oxidation of magnetite.

5. In boiling water and fog the addition of ammonia suppresses oxygen production without the formation of worrisome amounts of nitrate. At 5×10^{-4} M ammonia less than 0.01 cm^3 of oxygen and less than 0.01 mg

of nitrate are formed per kilogram of steam produced. When gases are continually stripped from the system, some ammonia decomposition occurs. When gases are not removed, equilibrium conditions are reached at moderate gas concentrations and net decomposition approaches zero.

CORROSION-PRODUCT TRANSPORT AND DEPOSITION

6. In systems made of carbon steel or stainless steel and cooled by pressurized water, boiling water or fog, the water contains filterable corrosion products at a concentration between 0.001 and 0.01 mg/kg. The amounts of colloidal and soluble corrosion products are generally in the same range. After several months in these systems Zircaloy fuel sheaths show deposits of corrosion products in the range from 10 to 200 $\mu g/cm^2$, all less than 1 μm in thickness.

RELEASE FROM DEFECTED UO_2 FUEL ELEMENTS

7. Irradiation of defected UO_2 fuel elements of different lengths operating at high power ratings has shown that gaseous and volatile fission products with half-lives less than a few hours are released to the coolant only from a small volume of fuel near the defect. Thus, release is controlled both by diffusion out of the UO_2 matrix and by slow movement along the fuel element to the defect.

8. Uranium released to the coolant deposits on system surfaces. Typical values on Zircaloy are 0.01 to 5 $\mu g\ U/cm^2$ depending on the size of defect and the time of exposure. The deposition is reversible and is related to the concentration of dissolved uranium in the coolant; however, rates of desorption are very low.

9. Power cycles cause the transient release of large amounts of fission-product activity. The resultant responses of delayed-neutron and gamma monitors are dependent on the rate and amount of the power increase, and can be used for locating defected fuel in a large reactor.

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Abstract

IRRADIATION EXPERIENCE WITH ZIRCALOY

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1. Zircaloy-2 is the alloy that was chosen for the pressure tubes and for the fuel cladding in both the NPD and CANDU reactors. Irradiations in the Chalk River research reactors and preliminary examinations of irradiated components from NPD indicate that the alloy's performance accords with the criteria established in the design of both pressure tubes and fuel.

2. Special equipment was designed and developed for measuring pressure tubes in NPD and in NRU loops. It has been shown that diametral creep is negligible and that the movement of fuel bundles causes no significant wear or fretting on the pressure-tube wall.

3. When in use, the pressure tubes are biaxially stressed, and their behaviour has been studied by testing to burst. The burst strength is higher than that calculated from measurements on specimens uniaxially stressed.

4. Fuel clad in Zircaloy-2 has successfully withstood irradiation to 10 000 MWd/tonne, and with hydrogen concentrations in the cladding of more than 200 ppm. Oxidation and hydriding of fuel cladding has sometimes been found to exceed that in equivalent samples not subject to the heat and radiation fluxes. Inferior surface conditions are thought to account, at least in part, for this. The amount of hydrogen absorbed by the sheath is 20 - 40% of the equivalent of the corrosion hydrogen.

5. Zircaloy-2 specimens with hydrogen concentrations up to 250 ppm have been irradiated to 10^{21} n/cm^2. The irradiation increases the tensile strength by 20 - 30% and decreases the elongation by 50%, but the various concentrations of hydrogen have negligible effect on the tensile properties. The impact strength decreases as a function of both neutron irradiation and hydrogen concentration.

6. The creep of cold-worked Zircaloy-2 has been measured in a fast-

neutron flux of 5 to 10 x 10^{12} cm^{-2} s^{-1} (E > 1 MeV) at 14 and 21 kgf/mm^2 and 300°C. In these tests it was observed that the in-reactor primary and early secondary creep was only slightly higher than in the laboratory, but the difference increased with test time. This was because the laboratory creep rates decreased continuously whereas in the reactor a steady rate was reached after approximately 1000 hours.

7. Zirconium hydrides precipitate parallel to the direction in which the grains are deformed and the degree of directionality increases with increasing strain and with heat-treatment temperatures up to 600°C. Annealing at temperatures between 600 and 800°C does not destroy directionality but decreases it. Below a strain of about 7% no directionality is observed.

8. Tensile tests have been carried out on specimens in which the stress was applied perpendicular to hydride platelets. The hydride did not have a marked effect on ductility unless the specimen thickness was decreased to about 10 times the length of the hydride platelets.

9. Corrosion samples have been subjected under irradiation to pressurized lithiated water, to ammoniated steam-water mixtures and to ammoniated superheated steam. Weight gains for these samples were similar to control samples in static autoclaves and not under irradiation. However, the hydriding rates were different and this may be due to different oxygen concentrations.

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Abstract

DEVELOPMENT OF ZIRCONIUM-NIOBIUM ALLOYS

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1. Data have been obtained which give considerable confidence in zirconium - 2.5wt% niobium alloys for use in water-cooled power reactors. A number of distinct advantages over the use of Zircaloy-2 can result.

2. The alloys studied, Zr - 2.5wt% Nb and Zr - 2.5wt% Nb - 0.5wt% Cu have contained ~1100 ppm oxygen as the major impurity. The best combination of strength, ductility and corrosion resistance found thus far is given by the heat treatments:

Zr - 2.5wt% Nb:	water quench from 850-880°C temper up to 24 h at 500°C
Zr - 2.5wt% Nb - 0.5wt% Cu:	water quench from 825-850°C temper 6 h at 535°C

The cold-worked condition is under investigation.

3. In the heat-treated condition, each alloy has an ultimate tensile strength at 300°C >56 kgf/mm^2, with a reduction in area of 70%. Neutron irradiation up to integrated fluxes of 10^{21} cm^{-2} (E >500 eV) has less deleterious effect on the ductilities of the zirconium-niobium alloys than on that of cold-worked Zircaloy-2.

4. Impact properties of the alloys have been determined both before and after irradiation, with hydrogen concentrations up to 250 ppm. Comparative studies between cold-worked Zircaloy-2 and the heat-treated zirconium-niobium alloys show the latter to absorb more energy at fracture.

5. The creep behaviour of the heat-treated Zr - 2.5wt% Nb alloy is characterized by a minimum rate of 5 x 10^{-6} %/h at 21 kgf/mm^2 and 300°C. Results on the Zr - 2.5wt% Nb - 0.5wt% Cu alloy indicate a creep rate slightly higher than for the binary alloy.

6. The corrosion behaviour has been evaluated for water, steam, air and carbon dioxide, at temperatures in the range 260-500°C. Post-transition weight gains in purified static water at 316°C are in the ranges 0.2-0.25 and 0.07-0.08 mg/(dm^2 day) for the Zr - 2.5wt% Nb and Zr - 2.5wt% Nb - 0.5wt% Cu alloys respectively. Corresponding hydrogen pick-up for the Zr - 2.5wt% Nb alloy is 2.5-4.5 μg/(dm^2 day). These values apply to material quenched from the β-phase and tempered 24 h at 500°C; similar rates are indicated when the alloys are in the preferred heat-treated conditions. Higher corrosion rates are found in flowing water at 288°C and pH 8-10 (adjusted by ammonia or lithium hydroxide). The corrosion behaviour of the Zr - 2.5wt% Nb - 0.5wt% Cu alloy in both air and moist carbon dioxide at 300°C is similar to that of Zircaloy-2; the corresponding behaviour for the Zr - 2.5wt% Nb alloy has been variable. The corrosion resistance of the Zr - 2.5wt% Nb alloy is improved by cold working between the quench and temper treatments.

7. A number of Zr - 2.5wt% Nb tubes, 5 m long with a 3-mm wall thickness, have been fabricated and heat-treated. These tubes have a burst strength at 300°C of >74 kgf/mm^2 with 15% reduction in area at point of rupture. Development programs are proceeding on fabrication of Zr - 2.5wt% Nb coolant tubes.

8. Coolant-tube spacers of heat-treated Zr - 2.5wt% Nb - 0.5wt% Cu alloy will be used in the CANDU reactor.

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Abstract

DEVELOPMENT OF FUEL AND COOLANT TUBES FOR A REACTOR COOLED BY ORGANIC LIQUID

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1. The first experimental reactor (WR-1) at Canada's Whiteshell Nuclear Research Establishment will be moderated by D_2O, cooled by organic liquid, and have vertical pressure tubes. The first pressure tubeswill be stainless steel and the first fuel will be UO_2. It is planned also to develop uranium carbide fuel and eventually to replace the pressure tubes using materials of low neutron absorption suitable for power reactors. This paper describes the development work on UO_2 fuels and on possible power-reactor pressure tubes.

2. The properties of SAP (sintered aluminum product) have been studied at high temperatures. Ultimate tensile strengths and total elongations have been measured on standard tensile specimens; axial creep, internal pressure creep and hoop stress rupture have been measured on tubes. The mechanical properties depend to a degree on the method of powder production and it may be possible to find improved methods of fabrication.

3. The fuel design is a 19-element rodded cluster with a rod separation of 1 mm, maximum sheath temperature of 500 °C, and fuel channel outlet temperature of 370 C. Sheathing materials are SAP and Zr-2.5%Nb. End closures in SAP have been achieved by pressure bonding and resistance welding and in Zr-2.5%Nb by shielded-arc and resistance welding.

4. Heat transfer to organic liquid has been studied under simulated reactor conditions and heat fluxes that cause departure from nucleate boiling (DNB) have been established. Local heat-transfer coefficients have been measured in open regions, in gaps between elements and in the vicinity of helical spacers in simulated fuel bundles. Those in the open regions agree with those measured in simple annular channels and are proportional to Re^n where $n > 0.8$. The DNB heat fluxes decrease with increasing test-section length up to 35 cm; beyond 35 cm they remain constant. For similar lengths they agree with data published by Atomics International.

5. Nine irradiation tests have been made on SAP-sheathed UO_2 fuels in the X-7 loop of the NRX reactor at $\int \lambda \, d\theta$ values of ~ 36 W/cm and sheath temperatures of 450 - 500°C. Four of these experiments resulted in sheath failures; two of these are considered to be significant. The significant failures occurred in SAP-930 (7% Al_2O_3). The mechanism of failure has not been established but there is some evidence that a stronger material such as SAP-895 (10%Al_2O_3) will be more resistant to failure. The results of an irradiation test made on UO_2 sheathed in Zr-2.5%Nb at a sheath temperature of 475°C are sufficiently encouraging to make Zr-2.5%Nb a contender for sheathing material.

6. Two coolant tube concepts are under development: externally insulated hot pressure tubes and internally insulated cold stress tubes. Hot pressure tubes of Zircaloy-4 have been tested in the X-7 loop; SAP hot pressure tubes are being tested in laboratory organic-coolant rigs. Zirconium-alloy cold stress tube assemblies, internally insulated with compressed silica fibre and lined with either SAP or zirconium-alloy tubes have performed well in laboratory tests. Irradiation tests are being made on the solid insulants to evaluate their stability. Techniques for mechanically joining SAP and zirconium-alloy tubes to ferrous metals have been developed and joints are under test.

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Abstract

CANADIAN RESERVES OF URANIUM AND THORIUM

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1. The rapid growth of the Canadian uranium industry in the post-war period will stand out as a remarkable achievement in Canadian mining history. But the recent decline in production, resulting from declining export markets, has been almost as rapid as the spectacular rise during the 1950's. At one time there were 25 mines and 19 treatment plants in operation, but now only one mine is expected to continue operating beyond 1967. Despite the severe contraction in production, virtually no loss of reserves will result from mine closures.

2. The exploration for uranium and thorium in Canada lasted for a very short time but, during the period 1950 to 1956, it resulted in the discovery of more than 10 000 radioactive occurrences among which was one of the largest known deposits in the world. Only a few of these occurrences reached the stage of production although a larger number were of considerable economic interest. The most important discoveries were the quartz-pebble conglomerates of the Blind River - Elliot Lake district of Ontario, the pitchblende-bearing vein deposits of the Beaverlodge Lake area in northern Saskatchewan and the pegmatitic granite dyke deposits near Bancroft in southeastern Ontario.

3. The present reserves of measured, indicated and inferred ore in Canada are 240×10^6 tonnes averaging 1.24 kg U_3O_8/tonne. This is equivalent to 222 000 tonnes of recoverable U_3O_8 or about 190 000 tonnes of uranium metal after deducting 20% for unrecoverable ore in pillars and 5% for ore-treatment losses. These reserves are based on a price for U_3O_8 ranging from $11 to about $17.50 a kilogram. Since 1953, a total of 57×10^6 tonnes of ore (containing 73 500 tonnes of U_3O_8) has been mined and milled. In addition to these reserves, the volume of lower-grade material, that is material ranging from 0.15 to 0.85 kg U_3O_8/tonne, is considered by the writers to be at least as large as the above-stated reserves. Most of this potential ore is in the Elliot Lake camp.

4. Occurrences of uranium and thorium are widespread, but most are in the Canadian Shield, particularly in its margins. Based on past experience in prospecting for uranium in Canada, and the extent of

unexplored areas which are considered to be geologically favorable, estimates of tonnages in possible future discoveries of ore and of resources of lower-grade material are given. The prospects of finding more ore are therefore considered to be favorable and additional prospecting should be done in the areas suggested using techniques explained in the paper.

5. Thorium is discussed also because it occurs with uranium in two of the principal deposits and is therefore inexpensive to recover as a by-product. Reasonably assured reserves of thorium "in place" are estimated at 145 000 tonnes of ThO_2. In addition to these reserves there are about 23 000 tonnes of ThO_2 in dumps of mill tailings.

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Abstract

LATTICE STUDIES AT CHALK RIVER AND THEIR INTERPRETATION

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1. Since the 1958 Geneva Conference, two new facilities for lattice tests have been brought into use at the Chalk River Nuclear Laboratories:

   - a zero-power D_2O-moderated lattice-test reactor (ZED-2) was commissioned in September 1960; it is 336 cm in diameter and 333 cm high.

   - a high-pressure high-temperature loop has been installed in the ZEEP reactor for studies of single rods at 280 °C.

2. The principal systems studied in ZED-2 have been

   - a charge of 7-element NPD bundles (each element is natural UO_2, density 10.2 g/cm^3, 2.40 cm in diameter and clad in 0.51 mm of aluminum).

   - a charge of 19-element CANDU bundles (each element is natural UO_2, density 10.45 g/cm^3, 1.42 cm in diameter and clad in 0.45 mm of Zircaloy-2).

   The CANDU bundles were surrounded by simulated pressure tubes and calandria tubes of aluminum. Zircaloy-2 pressure tubes have been used for some "few-rod" experiments. Measurements have been carried out over a range of hexagonal lattice spacings from 18 to 36 cm with coolants of D_2O, He and a hydrogenated polyphenyl whose approximate formula is $C_{18}H_{22}$.

3. The following lattice parameters have been measured (though not all for each lattice)

- buckling
- neutron density distribution in cells
- fast-fission ratio
- initial conversion ratio
- neutron temperature
- Westcott epithermal spectrum index (r)

4. Neutron spectrum parameters for 19-element CANDU bundles were also measured at high temperatures in the ZEEP loop with air and a range of H_2O-D_2O mixtures as coolants.

5. In addition, some measurements have been carried out with close-packed blanket-type lattices of uranium metal rods; the rods were both natural and depleted (0.26 wt%^{235}U), 3.25 cm in diameter and clad in 1.0 mm of aluminum. Measurements were carried out over a range of square lattice spacings from 4.06 cm to 8.13 cm with both D_2O and H_2O moderators.

6. The two-group calculation of lattice parameters described in 1958 has been continuously adjusted to fit new experimental data as they were obtained, and this adjustment is embodied in the POOOF computer program for reactor optimization studies. For more detailed and fundamental studies an advanced computer program, LATREP, is nearing completion and has been used for some comparisons with experimental results.

7. A second approach is the LATTICE ANALYSIS computer program which uses observed values obtained from the above intra-cell measurements to compute the basic parameters of the two-group four-factor criticality formula. This allows a detailed comparison between measurements and parameter values determined theoretically and, in principle, enables the determination of lattice buckling from intra-cell measurements.

8. Analytical treatments of slowing-down spectra and thermal-neutron spectra have been compared with measured values.

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FUEL BURN-UP AND REACTIVITY CHANGES

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1. A burn-up of about 10 000 MWd/tonne is predicted for natural uranium in large D_2O-moderated reactors. Until actual fuelling experience becomes available from operating reactors, the prediction of the reactivity limit on fuel burn-up is important.

2. Predictions from detailed calculations have been compared with measurements of reactivity change for uranium metal rods irradiated to about 6500 MWd/tonne. Agreement is reasonable considering the scatter of the experimental points and the uncertainties associated with epithermal absorption in fission products.

3. Measurements have been made of the reactivity loss in fission products. Samples of non-volatile fission products extracted from fuel irradiated to 6600 MWd/tonne and highly irradiated specimens of ^{233}U, ^{235}U and ^{239}Pu were measured for reactivity loss in a swimming-pool reactor (PTR). Burn-up of fissile material was determined by chemical and mass-spectrometric analyses. The results were compared with estimates of gross fission-product poisoning calculated from yields, cross-sections and resonance integrals. The comparison suggests that absorption of thermal neutrons was over-estimated in the calculations whereas the absorption of epithermal neutrons was underestimated.

4. The relative yields of ^{135}I in the fission of ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu were measured. The samples were irradiated in a high-flux reactor (NRU) for 24 hours and then the reactivity was followed in PTR as the ^{135}Xe concentration passed through its peak. The results show that the yield of ^{135}Xe in plutonium fission by this route is less than 6% and not the 7.4% assigned as the total yield of the 135 mass chain.

5. These experimental results provide a firm basis for calculating the fuel burn-up of an operating reactor. Initial calculations

of attainable fuel burn-up in a heavy-water power reactor (CANDU) used a two-group representation of the neutron spectrum. The variation of lattice parameters with burn-up was obtained from simple differential equations describing the changing fuel composition in an approximate way.

6. Subsequent calculations have been directed towards establishing the size and direction of errors in this simple first approach. Corrections have been made for the difference between the lattice recipes and experimental measurements on fuel assemblies similar to those proposed for CANDU. Approximations in the simple description of fuel change with irradiation have been examined by comparison with more detailed calculations. Effects peculiar to the bi-directional fuelling method have been studied in some detail. These effects include neutron spectrum change during irradiation, ^{239}Pu hold-up in ^{239}Np, and changes in xenon concentration, resonance escape probability and fuel temperature along a reactor channel.

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Abstract

A MEASUREMENT OF THE OCCURRENCE OF MINIMUM DILUTION RATES FOR STACK EFFLUENTS IN THE ATMOSPHERE

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1. When the rate of release of radioactive material to the atmosphere is divided by the concentration at ground level downwind, a coefficient K is obtained which is important in assessing the consequences of an accidental release of fission products from a nuclear reactor. If its value were known it could be used to specify the siting and containment provisions necessary in order that the reactor might not be a hazard to the population. K varies widely between infinity and some minimum value depending on the wind direction, weather conditions and the time period of the release. When calculating the value of K to be used in assessing the consequences of an accident it is difficult to select numbers for the parameters in the diffusion equation, although it is usual to assume that there is an inversion during the release. The resulting values of K may be so pessimistic as to be unrealistic because the postulated diffusion conditions occur so rarely or persist for very short periods.

2. This work was undertaken to determine, from operating experience in an area surrounding a stack, the lowest values of K as well as their frequency and duration. The tracer used was ^{41}Ar, produced during passage of cooling air through the NRX and NRU reactors and discharged through a single stack at the Chalk River Nuclear Laboratories. These Laboratories lie by the Ottawa River on terrain which is typical of that surrounding many inland power-reactor sites. The results of these studies should therefore be applicable to other locations.

3. Four sampling stations were established along a line parallel to the river valley and also to the direction of the prevailing winds. Three were located in one direction at 1.5, 4.8 and 9.6 km respectively from the stack and the fourth at 1 km in the diametrically opposite direction. The results could therefore be used to determine the effect of distance on the values of K and also on the probability of their occurrence.

4. In the first half of 1963 the lowest values of K recorded at each of the four sampling points were not significantly different from one another (average about 4×10^4 m^3/s). The effect of distance from the point of release was, however, to reduce markedly the probability of occurrence of such low values. Thus for values of K in the range 3 to 10×10^4 m^3/s, the percentage incidence decreased from about 2% at the nearest to about 0.01% at the furthest location.

5. The results so far obtained have been used in Canada to assess the containment provisions necessary for the Nuclear Power Demonstration reactor at Rolphton, Ontario and the experimental reactor, WR-1, to be built at Whiteshell, Manitoba.

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REACTOR SAFETY PRACTICE AND EXPERIENCE IN CANADA

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1. In Canada, the Atomic Energy Control Board is the regulatory agency in the field of atomic energy. It discharges its responsibilities for reactor safety with the assistance and advice of national and provincial departments of health and of labour, and of a Reactor Safety Advisory Committee. Canadian thinking and practice in regard to reactor safety has been greatly influenced by experience and research in the laboratories of Atomic Energy of Canada Limited.

2. Experience and experiment have led to such principles as: the "safety bank" or shutdown reserve, the use of redundancy in control systems to permit quick discovery and correction of faults, the use of moderator-level control with associated dump for emergency shutdown, the independence of regulatory and protective systems, the need for two independent trips for any possible control failure, provision of adequate emergency coolant for the fuel, and the requirement that emergency or stand-by devices should not remain idle or untested for long periods. The extension of some of these principles since the Second Geneva Conference is described.

3. In appraising the hazards of reactors, greater consideration is given to the probability of failures of components of the equipment that might lead to grave consequences, than to the number of casualties and other consequences of the "worst credible accident", because it directs attention to the causes of accidents and thereby promotes improvement in safety. Increasing attention is being given to determining unreliability of reactor equipment by observation of actual operation.

4. It is considered especially important that the process systems (including reactivity regulation), the automatic protective devices, and the containment provisions, be independent of one another so that the simultaneous failure of any two of them from the same cause is very improbable. The automatic protective equipment should be such that

any failure in the regulating system would not be expected to result in injury to any person inside or outside the plant. The containment provisions should be designed to prevent injury to persons outside the plant area if the regulating system, or other process equipment, and the automatic protective devices fail simultaneously. All equipment should be designed so that faults in parts or components can be discovered soon after they occur and corrected without delay.

5. Considerable experimental and development work has been done on containment provisions such as vapour suppression by water spray, and the automatic closure of the containment shell following a loss-of-coolant accident before the fuel can be overheated by decaying fission-product activity. Other safety problems that have been investigated experimentally include: damage from the accidental bursting of a coolant pressure tube in the reactor, the overheating of fuel elements, the behaviour of gaseous fission products in overheated oxide fuel, and the dispersal of fission products, if accidentally released, in the atmosphere, in the ground, and in rivers and lakes.

6. Advances in reactor safety and continuing experience in safe operation now encourage the location of reactors in somewhat more populated regions than formerly.

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THE MANAGEMENT OF HEAVY WATER FOR RESEARCH AND POWER REACTORS

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1. Canada has had experience with several heavy-water reactor systems including one operating at high pressure and high temperature. All the systems are complex and the larger ones allow entry for changing fuel with the reactor at power. Inevitably some heavy water escapes and the systems have been designed to keep the quantities that escape to a minimum. Tritium is associated with any irradiated heavy water and the concentrations that arise when there is an escape call for measures to prevent hazards to personnel.

2. During the last six years the amount of H_2O entering the D_2O systems of the NRX and NRU reactors was less than 100 parts per million per year. The major problem was the reclamation of heavy water that escaped from the reactors, particularly when it was diluted with H_2O (i.e. downgraded) or contaminated with undesirable impurities. The amount of downgraded heavy water collected during routine reactor operation was not large (500 kg/year for the NRX reactor), but, in three instances, very large quantities (>10 000 kg) of heavy water were downgraded and in each instance the reclamation problems were different. The recovery of D_2O from ventilation systems is usually not economic, but a commercial unit with alumina has been effective in a ventilation system at the Nuclear Power Demonstration reactor (NPD). Most impurities, with the exception of organics, are removed successfully on ion-exchange resin. Oil can be brought down to about one part per million with filter-aids and then this trace removed with an ion-exchange resin.

3. Distillation and electrolysis have both been used for the isotopic reconcentration of water containing less than 99.75 wt% D_2O. Distillation is better where the full benefits of an automated system can be realized, i.e. for removing small quantities of H_2O from large volumes of D_2O and for reconcentrating large volumes of low-grade water (< 65 wt% D_2O).

The electrolytic process is more versatile and is particularly well suited to water containing over 90 wt% D_2O; its small batch size, as low as 50 kg for a single cell, is an advantage. The separation factor α, which is a measure of the efficiency of an electrolytic plant, usually exceeded nine and often exceeded ten.

4. The isotopic analysis of water containing less than 99 wt% D_2O is not accurate enough for precise accounting, especially when large volumes are involved.

5. The tritium concentration of the D_2O in the NRX and NRU reactors has been about 5 curies/liter for several years. From January to October, 1963, 295 persons were exposed to tritium; of these 87% received less than 0.5 rem and less than 1% exceeded 1 rem. All the operators in the electrolytic plant received less than 0.5 rem. Adequate ventilation, good housekeeping, properly designed equipment, and sound operating procedures are key factors in the control of this hazard.